NUCLEAR REGULATORY COMMISSION

[Docket No. 50–382]

Entergy Operations, Inc. (Waterford Steam Electric Station, Unit No. 3); Exemption

I

Entergy Operations, Inc., (the licensee) is the holder of Facility Operating License No. NPF–38, which authorizes operation of Waterford Steam Electric Station Unit No. 3 (the facility, Waterford 3). The operating license provides among other things, that it is subject to all rules, regulations, and orders of the Commission now or hereafter in effect. The facility is a pressurized water reactor located at the licensee's site in St. Charles Parish, Louisiana.

II

Section III.D.1.(a) of Appendix J to 10 CFR Part 50 requires the performance of three Type A containment integrated leakage rate tests (ILRTs), at approximately equal intervals during each 10-year service period of the primary containment.

III

By letter dated November 16, 1993, as supplemented by letters dated August 19, 1994, March 30, and June 19, 1995, the licensee requested temporary relief from the requirement to perform a set of three Type A tests at approximately equal intervals during each 10-year service period of the primary containment. The requested exemption would permit a one-time interval extension of the third Type A test by approximately 18 months (from the 1995 refueling outage, currently scheduled to begin in September 1995, to the 1997 refueling outage).

The licensee's request primarily cites the special circumstances of 10 CFR 50.12, paragraph (a)(2)(ii), as the basis for the exemption. They point out that the existing Type B and C testing programs are not being modified by this request and will continue to effectively detect containment leakage caused by the degradation of active containment isolation components as well as containment penetrations. The licensee also indicated that the testing history, structural capability of the containment, and the risk assessment has established that Waterford 3 has a low leakage containment, the structural integrity of the containment is assured, and that there is a neglible risk impact in changing the Type A test schedule. Therefore, application of the regulation in this particular circumstance would not serve, nor is it necessary to achieve, the underlying purpose of the rule.

IV

Section III.D.1.(a) of Appendix J to 10 CFR Part 50 states that a set of three Type A leakage rate tests shall be performed at approximately equal intervals during each 10-year service period.

The licensee proposes an exemption to this section which would provide a one-time interval extension for the Type A test by approximately 18 months. The Commission has determined, for the reasons discussed below, that pursuant to 10 CFR 50.12(a)(1) this exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. The Commission further determines that special circumstances, as provided in 10 CFR 50.12(a)(2)(ii), are present justifying the exemption; namely, that application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule.

The underlying purpose of the requirement to perform Type A containment leak rate tests at intervals during the 10-year service period, is to ensure that any potential leakage pathways through the containment boundary are identified within a time span that prevents significant degradation from continuing or becoming unknown. The NRC staff has reviewed the basis and supporting information provided by the licensee in the exemption request. The NRC staff has noted that the licensee has a good record of ensuring a leak-tight containment. All Type A tests have passed with significant margin and the licensee will continue to perform the existing Type B and C testing to detect containment leakage caused by the degradation of active containment isolation components as well as containment penetrations. The licensee has stated to the NRC Project Manager that they will perform the general containment inspection although it is only required by Appendix J (Section V.A.) to be performed in conjunction with Type A tests. The NRC staff considers that these inspections, though limited in scope, provide an important added level of confidence in the continued integrity of the containment boundary.

The NRC staff has also made use of the information in a draft staff report, NUREG–1493 ''Performance-Based Containment Leak-Test Program,'' which provides the technical justification for the present Appendix J rulemaking effort which also includes a 10-year test interval for Type A tests. The integrated leakage rate test, or Type A test, measures overall containment leakage. However, operating experience with all types of containments used in this country demonstrates that essentially all containment leakage can be detected by local leakage rate tests (Type B and C). According to results given in NUREG–1493, out of 180 ILRT reports covering 110 individual reactors and approximately 770 years of operating history, only 5 ILRT failures were found which local leakage rate testing could not detect. This is 3% of all failures. This study agrees well with previous NRC staff studies which show that Type B and C testing can detect a very large percentage of containment leaks.

The Nuclear Management and Resources Council (NUMARC), now the Nuclear Energy Institute (NEI), collected and provided the NRC staff with summaries of data to assist in the Appendix J rulemaking effort. NUMARC collected results of 144 ILRTs from 33 units; 23 ILRTs exceeded $1.0L_a$. Of these, only nine were not due to Type B or C leakage penalties. The NEI data also added another perspective. The NEI data show that in about one-third of the cases exceeding allowable leakage, the as-found leakage was less than $2L_a$; in one case the leakage was found to be approximately $2L_a$; in one case the as-found leakage was less than $3L_a$; one case approached $10L_a$; and in one case the leakage was found to be approximately $21L_a$. For about half of the failed ILRTs the as-found leakage was not quantified. These data show that, for those ILRTs for which the leakage was quantified, the leakage values are small in comparison to the leakage value at which the risk to the public starts to increase over the value of risk corresponding to L_a (approximately $200L_a$, as discussed in NUREG–1493). Therefore, based on these considerations, it is unlikely that an extension of one cycle for the performance of the Appendix J, Type A test at Waterford 3 would result in significant degradation of the overall containment integrity. As a result, the application of the regulation in these particular circumstances is not necessary to achieve the underlying purpose of the rule.

Based on generic and plant specific data, the NRC staff finds the basis for the licensee's proposed exemption to allow a one-time exemption to permit a schedular extension of one cycle for the performance of the Appendix J, Type A test to be acceptable provided the general containment inspection (10 CFR Part 50, Appendix J, Section V.A.) is performed.

Pursuant to 10 CFR 51.32, the Commission has determined that granting this Exemption will not have a significant impact on the environment (60 FR 39020).

This Exemption is effective upon issuance and shall expire after March 31, 1997, or at the completion of the 1997 refueling outage whichever comes first.

Dated at Rockville, Maryland, this 3rd day of August 1995.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

Deputy Director, Division of Reactor Projects III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 95–20027 Filed 8–11–95; 8:45 am]

BILLING CODE 7590–01–P

[Docket No. 50–315]

Indiana Michigan Power Co.; Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License No. DPR–58, issued to Indiana Michigan Power Company (the licensee), for operation of the Donald C. Cook Nuclear Plant, Unit 1, located in Berrien County, Michigan.

The proposed amendment would modify technical specifications 4.4.5.4 and 4.4.5.5, on steam generators, to allow for repair of hybrid expansion joint sleeves under redefined repair boundary limits.

The licensee requested this change on an exigent basis because: (1) The change is associated with steam generator tube repairs during the Unit 1 refueling outage currently in progress, and (2) the empirical data compiled from the Kewaunee Nuclear Plant steam generator tube pulls in March 1995 is the primary support for this amendment and the final implications and conclusions from assessment of that data are just now being formulated. The Unit 1 tube repairs are currently scheduled to begin on August 29, 1995.

The NRC staff has reviewed and concurred with the licensee's reasons for requesting this amendment on an exigent basis.

Before issuance of the proposed license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations.

Pursuant to 10 CFR 50.91(a)(6), for amendments to be granted under exigent circumstances the NRC staff must determine that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) Operation of the CNP [Donald C. Cook Nuclear Plant] unit 1 in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Mechanical testing has shown that the inherent structural strength of the HEJ [hybrid expansion joint] provides sufficient integrity such that the tube rupture capability recommendations of RG [Regulatory Guide] 1.121 are met, even for instances of 100% throughwall, 360° circumferentially oriented degradation in the HEJ hardroll lower transition region. Structural integrity recommendations consistent with RG 1.121 are supplied for all tube degradation 1.1 inch or greater below the bottom of the HEJ hardroll upper transition. Based on test data, a bounding SLB [steam line break] leak rate of 0.033 gpm for indications between 1.1 and 1.3 inch below the bottom of the hardroll upper transition is applied. As the leakage data base is expanded and statistical basis established, this SLB leakage allowance may be reduced. For indications existing greater than 1.3 inch below the bottom of the hardroll upper transition, SLB event leakage can be neglected.

Additional prevention from tube rupture is inherently provided by the HEJ geometry. For RCS [reactor coolant system] release rates to exceed the normal makeup capacity of the plant, approximately 120 gpm, the tube must be postulated to experience a complete circumferential separation at the lower transition, and become axially displaced by 3 to 3.25 inches, resulting in complete geometric disassociation between the tube and sleeve resulting in sufficient flow area to support leakage of 120 gpm. During the 1989 plug top release event at North Anna unit 1, primary to secondary release rates were calculated to be less than 80 gpm, for a flow area approximately 4 times larger than the flow area created by a tube which has axially displaced by about 1.25 to 1.5 inch. Analysis of the steam generator indicates that at a 95% cumulative probability, the tube would experience an axial displacement of less than the 1.1 inch boundary. At this level of axial displacement, a ring of metal to metal contact would remain between the tube and sleeve, and leakage would be far less than 120 gpm. Projected leakage at this point is expected to be less than 2.5 gpm. Therefore, implementation of the proposed repair boundary will not result in tube rupture, even for a tube postulated to not behave as predicted by the available test and pulled tube data.

The proposed technical specification change to support the implementation of the HEJ sleeve tube repair boundary for parent tube degradation in the HEJ hardroll lower transition region does not adversely impact any other previously evaluated design basis accident or the results of accident analyses for the current technical specification minimum reactor coolant system flow rate. Plugging limit criteria are established using the guidance of RG 1.121. Furthermore, per RG 1.83 recommendations, the sleeved tube assembly can be monitored through periodic inspections with present eddy current techniques.

(2) The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Implementation of the repair boundary will not introduce significant or adverse changes to the plant design basis. Mechanical testing of degraded sleeve joints supports the conclusions of the calculations that the sleeve retains structural (tube burst) capability consistent with RG 1.121. As with [the] initial installation of sleeves, implementation of the alternate criteria cannot interact with other portions of the RCS. Any hypothetical accident as a result of potential tube degradation in the HEJ hardroll lower transition region of the tube is bounded by the existing tube rupture accident analysis. Neither the sleeve design nor implementation of the tube repair boundary defined in Attachment 4 [Westinghouse Electric Corporation Proprietary Report, WCAP–14446] affects any other component or location of the tube outside of the immediate area repaired. In addition, as the installation of sleeves and the impact on current plugging level analyses is accounted for, any postulation that the alternate repair criteria for parent tube degradation in the HEJ hardroll lower transition creates a new or different type of accident is not supported.

(3) The proposed license amendment does not involve a significant reduction in a margin of safety.

The safety factors used in the establishment of the HEJ sleeved tube alternate repair boundary for the disposition of indications in the hardroll lower transition of potentially degraded parent tubes are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used in steam generator design. Based on the sleeved tube geometry, it is unrealistic to consider that application of the repair boundary could result in single tube leak